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Thomas G. Scarbrough

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PERIODIC VERIFICATION OF DESIGN-BASIS CAPABILITY OF SAFETY-RELATED MOTOR-OPERATED VALVES

Thomas G. Scarbrough
Mechanical and Civil Engineering Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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Abstract

The safe operation of a nuclear power plant depends on motor-operated valves (MOVs) in fluid systems successfully performing their safety functions. As a result of problems with MOV performance, the NRC issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of the Design-Basis Capability of Safety-Related Motor-Operated Valves," requesting that nuclear power plant licensees verify initially and periodically the design-basis capability of MOVs in safety-related systems. The NRC also issued GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," requesting that licensees ensure that safety-related power-operated gate valves susceptible to pressure locking or thermal binding are capable of performing their safety functions. Licensees of all active operating reactor units have completed their programs to verify initially the design-basis capability of safety-related MOVs in response to GL 89-10, and to address potential pressure locking and thermal binding of safety-related power-operated valves in response to GL 95-07. In response to GL 96-05, the owners groups developed an industry-wide Joint Owners Group (JOG) program for periodic verification of the design-basis capability of safety-related MOVs. Most licensees committed to implement the JOG program as part of their response to GL 96-05. The NRC staff reviewed the establishment of GL 96-05 programs at individual nuclear plants through significant reliance on licensee commitments to implement the JOG program on MOV periodic verification. JOG has completed its MOV dynamic testing program, and prepared its topical report for use by licensees in implementing their MOV periodic verification programs. The NRC staff is currently reviewing the JOG final topical report. This paper provides an update of the NRC staff activities regarding the periodic verification of the design-basis capability of safety-related MOVs, and monitoring of the industry's efforts to ensure proper performance of safety-related MOVs.

I. INTRODUCTION

The safe operation of a nuclear power plant depends on motor-operated valves (MOVs) in fluid systems successfully performing their safety functions. MOVs must be capable of operating under design-basis conditions, which may include high differential pressure and flow, high ambient temperature, and degraded motor voltage. The design of the MOV must apply valid engineering equations and parameters to ensure that the MOV will operate as intended during normal plant operations and design-basis events. Manufacturing, installation, preoperational testing, operation, inservice testing (IST), maintenance, and replacement must be conducted by trained personnel using proper procedures. Surveillance must be performed and testing criteria must be applied on a soundly based frequency in a manner that suitably detects questionable operability or degradation. Moreover, these activities must be monitored by a strong quality assurance program.

The regulations of the U.S. Nuclear Regulatory Commission (NRC) require that components that are important to the safe operation of a U.S. nuclear power plant be treated in a manner that ensures their performance. Appendix A, "General Design Criteria for Nuclear Power Plants," and Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) contain broadly based requirements in this regard. In 10 CFR 50.55a, the NRC initially required U.S. nuclear power plant licensees to implement provisions of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (B&PV Code) for testing of MOVs as part of their

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IST programs. In 1999, the NRC revised 10 CFR 50.55a to incorporate by reference the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) for inservice testing of MOVs. The NRC also supplemented the quarterly MOV stroke-time testing specified in the ASME Code by requiring that licensees verify MOV design-basis capability on a periodic basis.

Operating experience at nuclear power plants in the 1980s and 1990s revealed weaknesses in many activities associated with MOV performance. For example, some engineering analyses used in the original sizing and setting of MOVs did not adequately predict the thrust and torque required to open and close valves under design-basis conditions. Both regulatory and industry research programs later confirmed the weakness in the initial design and qualification of MOVs. For example, the NRC Office of Nuclear Regulatory Research sponsored an extensive program at the Idaho National Engineering and Environmental Laboratory (INEEL) to study the performance of MOVs under various flow, temperature, and voltage conditions. In addition, the nuclear industry sponsored a significant program by the Electric Power Research Institute (EPRI) to develop a computer methodology to predict the performance of MOVs under a wide range of operating conditions. Poor MOV performance also resulted from shortcomings in maintenance programs, such as inadequate procedures and training. Further, testing of MOVs to measure valve stroke times under zero differential-pressure and flow conditions was shown not to detect certain deficiencies that could prevent MOVs from performing their safety functions under design-basis conditions.

II. VERIFICATION OF MOV DESIGN-BASIS CAPABILITY

In response to weaknesses in MOV performance, the NRC staff issued Generic Letter (GL) 89-10 (June 28, 1989), "Safety-Related Motor-Operated Valve Testing and Surveillance." In GL 89-10, the NRC staff requested that licensees ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. The NRC staff requested that licensees complete their GL 89-10 programs within approximately three refueling outages or 5 years of the issuance of the generic letter.

In support of the regulatory activities to ensure MOV design-basis capability, the NRC Office of Nuclear Regulatory Research identified areas in which research and analysis were required to assist in evaluating MOV programs at nuclear power plants. For example, the NRC performed research to evaluate (1) performance of MOVs under pump flow and blowdown conditions; (2) output of ac-powered and dc-powered MOV motor actuators; (3) the increase in friction of aged samples of valve materials; (4) methods to determine appropriate values for stem friction coefficient; (5) pressure locking and thermal binding of gate valves; and (6) the effect of ambient temperature on stem lubricant performance. The NRC sponsored flow testing of several MOVs by INEEL under normal flow and blowdown conditions. The testing revealed that (1) more thrust was required to operate gate valves than predicted by standard industry methods; (2) some valves were internally damaged under blowdown conditions and their operating requirements were unpredictable; (3) static and low flow testing might not predict valve performance under design-basis flow conditions; (4) during valve opening strokes, the highest thrust requirements might occur at unseating or in the flow stream; (5) partial valve stroking did not reveal the total thrust required to operate the valve; (6) torque, thrust, and motor operating parameters were needed to fully characterize MOV performance; and (7) reliable use of MOV diagnostic data requires accurate equipment and trained personnel. The NRC provided detailed test results in NUREG/CR-5406 (October 1989), "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test;" NUREG/CR-5558 (January 1991), "Generic Issue 87: Flexible Wedge Gate Valve Test Program;" NUREG/CR-5720 (June 1992), "Motor-Operated Valve Research Update;" and NUREG/CR-6100 (September 1995), "Gate Valve and Motor-Operator Research Findings." The NRC summarizes some of the results of the MOV research program in NRC Information Notice (IN) 90-40 (June 5, 1990), "Results of NRC-Sponsored Testing of Motor-Operated Valves." Additional examples of MOV research sponsored by the NRC are discussed later in this paper.

To assist nuclear power plant licensees in responding to GL 89-10, EPRI developed the MOV Performance Prediction Methodology (PPM) to determine dynamic thrust and torque requirements for gate, globe, and butterfly valves based on first-principles of MOV design and operation. EPRI described the methodology in Topical Report TR-103237 (Revision 2, April 1997), "EPRI MOV Performance Prediction Program." The EPRI MOV PPM program included the development of improved methods for prediction and evaluation of system flow parameters; gate, globe, and butterfly valve performance; and motor-actuator rate-of-loading effects (load sensitive behavior). EPRI also performed separate effects testing to provide information for refining the gate valve model and rate-of-loading methods; and conducted numerous MOV tests to provide data for development and validation of the models and methods, including flow loop testing, parametric flow loop testing of butterfly valve disk designs, and in-situ MOV testing. EPRI integrated the individual models and methods into an overall methodology including a computer model and implementation guide. On March 15, 1996, the NRC staff issued a safety evaluation (SE) accepting the EPRI MOV PPM with certain conditions and limitations. On February 20, 1997, the staff issued a supplement to the SE on general issues and two unique gate valve designs. On April 20, 2001, the staff issued Supplement 2 to the SE on Addendum 1 to EPRI Topical Report TR-103237 addressing an update of the computer model.

On September 8, 1999, the Nuclear Energy Institute (NEI) submitted Addendum 2 to EPRI Topical Report TR-103237-R2, which described the development of the Thrust Uncertainty Method that takes into account conservatism in the EPRI MOV PPM to provide a more realistic (less bounding) estimate of the thrust required to operate gate valves than predicted by the PPM. In this effort, EPRI compared the thrust required to operate sample gate valves during flow loop tests conducted as part of the development of the PPM to the thrust requirement predicted by the PPM to establish a representative prediction ratio for the actual-to-predicted thrust required to operate the valves. In applying the Thrust Uncertainty Method, a licensee would use the representative prediction ratio to reduce the EPRI MOV PPM thrust prediction for a specific gate valve to a nominal value. The licensee would determine a thrust prediction uncertainty for that valve based on the EPRI MOV PPM thrust prediction and the nominal thrust prediction obtained using the Thrust Uncertainty Method. The licensee would then establish a minimum thrust to be provided at the control switch trip setpoint (or flow isolation) for the applicable MOV, based on the nominal thrust prediction of the Thrust Uncertainty Method combined with applicable bias and random setup uncertainties (including rate-of-loading effects, diagnostic test equipment uncertainty, control switch repeatability, and the thrust prediction uncertainty). In Supplement 3 (dated September 30, 2002) to the SE on the EPRI PPM, the NRC staff concluded that the Thrust Uncertainty Method developed by EPRI is acceptable for the prediction of minimum allowable thrust at control switch trip (or flow isolation) for applicable motor-operated gate valves under cold water applications within the scope of the Thrust Uncertainty Method, based on the NRC staff's review of Addendum 2 to the EPRI Topical Report as supplemented by NEI submittals dated January 5 and December 6, 2001, and June 10, 2002. Therefore, the NRC staff stated that the Thrust Uncertainty Method may be applied consistent with the criteria specified for the EPRI MOV PPM in EPRI TR-103237-R2 and Addenda 1 and 2 to TR-103237-R2, as supplemented by NEI submittals dated January 5 and December 6, 2001, and June 10, 2002. The NRC staff noted that its findings and conclusions on the use of the EPRI MOV PPM, and applicable limitations and conditions, are provided in the SE dated March 15, 1996; the SE supplements dated February 20, 1997; April 20, 2001; and September 30, 2002.

NRC Information Notice (IN) 96-48 (August 21, 1996), "Motor-Operated Valve Performance Issues," alerted licensees to lessons learned from the EPRI MOV program. Among the lessons learned were: (1) the thrust requirements to operate some gate valves under pump flow and blowdown conditions were higher than predicted by the valve manufacturers; (2) a potential exists for gate valves to be damaged when operating under blowdown conditions such that the thrust requirements can be unpredictable; (3) the effective flow area in some globe valves can be larger than expected and can cause thrust requirements to be higher than predicted; and (4) the friction coefficients for sliding surfaces in gate valves can increase with service before reaching a plateau. In IN 96-48, the staff noted that some of the EPRI information is applicable to gate, globe, and butterfly valves regardless of the type of actuator operating the valve.

Nuclear power plant licensees implemented the recommendations of GL 89-10 through a combination of design-basis reviews, revision of MOV calculations and procedures, static and dynamic diagnostic testing, industry-sponsored research programs, and trending of test results. The industry expended significant resources to resolve the deficiencies in the design, qualification, and application of safety-related MOVs that led to the issuance of GL 89-10. The results of the GL 89-10 programs and their implementation include (1) MOV sizing calculations and switch settings have been revised to reflect actual valve performance; (2) improved valve performance prediction methods have been developed; (3) valve internal dimensions are being addressed to provide assurance of predictable gate valve performance under blowdown conditions; (4) friction coefficients in new or refurbished gate valves have been found to increase with service until a plateau reached; (5) MOV output prediction methods have been updated; and (6) personnel training and maintenance practices have been improved. The NRC staff has evaluated the MOV program at each nuclear plant through onsite inspections of the design-basis capability of safety-related MOVs. The NRC staff has closed its review of GL 89-10 for each active U.S. nuclear power plant.

III. LONG-TERM ASPECTS OF MOV PERFORMANCE

On September 18, 1996, the NRC staff issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," to provide recommendations for assuring the capability of safety-related MOVs to perform their design-basis functions over the long term. In GL 96-05, the NRC staff requested that licensees establish a program, or ensure the effectiveness of their current program, to verify on a periodic basis that safety-related MOVs continue to be capable of performing their safety functions within the current licensing basis of the facility. The guidance in GL 96-05 supersedes the guidance in GL 89-10 on long-term MOV programs.

In GL 96-05, the NRC staff noted five attributes of effective programs for periodic verification of safety-related MOV design-basis capability at nuclear power plants:

(1) A risk-informed approach may be used to prioritize valve test activities, such as frequency of individual valve tests and selection of valves to be tested.

(2) The valve test program provides adequate confidence that safety-related MOVs will remain operable until the next scheduled test.

(3) The importance of the valve is considered in determining an appropriate mix of exercising and diagnostic testing. In establishing the mix of testing, the benefits (such as identification of decreased thrust output and increased thrust requirements) and potential adverse effects (such as accelerated aging or valve damage) are considered when determining the appropriate type of periodic verification testing for each safety-related MOV.

(4) All safety-related MOVs covered by the GL 89-10 program are considered in the development of the periodic verification program. The program includes safety-related MOVs that are assumed to be capable of returning to their safety position when placed in a position that prevents their safety system (or train) from performing its safety function; and the system (or train) is not declared inoperable when the MOVs are in their nonsafety position.

(5) Valve performance and maintenance are evaluated and monitored, and the periodic verification program is periodically adjusted as appropriate.

JOG Program on MOV Periodic Verification

In response to GL 96-05, nuclear power plant owners groups developed an industry-wide Joint Owners Group (JOG) Program on MOV Periodic Verification to obtain benefits from sharing information between licensees on MOV performance. Elements of the JOG program included (1) an "interim" MOV periodic verification program for applicable licensees to use in response to GL 96-05; (2) a 5-year dynamic testing program to identify potential age-related increases in required thrust and torque to operate gate, globe, and butterfly valves under dynamic conditions; and (3) a long-term MOV diagnostic program based on information from the dynamic testing program. On October 30, 1997, the NRC staff issued an SE accepting the JOG Program on MOV Periodic Verification with certain conditions and limitations.

Licensees of 98 reactor units have participated in the JOG program. The JOG 5-year dynamic testing program included 176 valves that received three dynamic tests with at least a 1-year time interval between the tests. An additional 14 valves received two dynamic tests with at least a 1-year time interval between the tests. In total, the JOG program included 514 dynamic valve tests and involved 52 person-years of effort. The JOG program constituted the largest set of MOV dynamic tests obtained to date for use by U.S. nuclear power plant licensees.

One of the key observations from the JOG program was that an increase in the required thrust or torque did not occur due only to the passage of time (without operation of the valve under dynamic fluid conditions). Further, the JOG program results indicated that significant service-related degradation in valve performance is not expected for MOVs as currently designed, installed and maintained in nuclear power plants. However, the MOV tests revealed that, where the initial valve factor is low because of prior disassembly of the valve or its limited service under dynamic fluid conditions, the thrust requirements for gate valves can increase significantly up to a bounding value over their service life. The program also found that a significant variation can occur in the operating torque requirements for butterfly valves with bronze bearings without a hub seal installed in untreated water systems; and for butterfly valves with non-metallic bearings.

On February 27, 2004, the JOG submitted Topical Report MPR-2524 (Revision 0, February 2004), "Joint Owners' Group (JOG) Motor Operated Valve Periodic Verification Program Summary," providing the long-term recommendations for MOV periodic verification to be implemented by licensees as part of their commitments to GL 96-05. The NRC staff plans to prepare an SE on its evaluation of the JOG topical report. The NRC staff hopes to complete the SE later in 2004.

Owners Group's MOV Risk Categorization Methodologies

Licensees are applying risk insights in implementing their long-term MOV programs. In Topical Report NEDC 32264, "Application of Probabilistic Safety Assessment to Generic Letter 89-10 Implementation," the Boiling Water Reactor Owners' Group (BWROG) describes a methodology to rank MOVs according to their relative importance to core damage frequency and other considerations to be applied by an expert panel. On February 27, 1996, the NRC staff issued an SE accepting the BWROG methodology for risk ranking MOVs with certain conditions and limitations. On June 2, 1997, the Westinghouse Owners' Group (WOG) submitted Engineering Report V-EC-1658 (Revision 1) describing an MOV risk-ranking approach for Westinghouse-design nuclear plants. On April 14, 1998, the NRC staff issued an SE accepting the WOG methodology for risk ranking MOVs with certain conditions and limitations.

Performance of ac-Powered MOV Actuators

In that the JOG program focused on potential increases in valve operating requirements, licensees address potential degradation in the output of MOV motor actuators by their plant-specific programs. In the late 1990s, the NRC sponsored research at INEEL to study the performance of ac-powered MOV motor actuators manufactured by Limitorque Corporation, under various temperature and voltage conditions. For the Limitorque ac-powered motor-actuator combinations tested, the research indicated that (1) actuator efficiency might not be maintained at "run" efficiency published by the manufacturer; (2) degraded voltage effects can be more severe than predicted by the square of the ratio of actual to rated motor voltage; (3) some motors produce more torque output than predicted by their nameplate rating; and (4) temperature effects on motor performance appeared consistent with the Limitorque guidance. The NRC study of ac-powered MOV output is described in NUREG/CR-6478 (July 1997), "Motor-Operated Valve (MOV) Actuator Motor and Gearbox Testing." The nuclear industry also evaluated the output capability of ac-powered MOVs at several plants. In response to the new information on ac-powered MOV performance, Limitorque provided updated guidance in its Technical Update 98-01 (May 15, 1998) and Supplement 1 (July 17, 1998) for the prediction of ac-powered MOV motor actuator. The NRC alerted licensees to the new information on ac-powered MOV output in Supplement 1 (July 24, 1998) to IN 96-48.

Performance of dc-Powered MOV Actuators

Following the NRC review of ac-powered MOV performance, the NRC sponsored research at INEEL to study the performance of Limitorque dc-powered MOV motor actuators under various temperature and voltage conditions. For the Limitorque dc-powered motor-actuator combinations tested, the research indicated that (1) ambient temperature effects were more significant than predicted; (2) use of a linear voltage factor needs to consider reduced speed, increased motor temperature, and reduced motor output; (3) stroke-time increase is significant for some dc-powered MOVs under loaded conditions; and (4) actuator efficiency may fall below the published "pullout" efficiency at low speed and high load conditions. The research results are provided in NUREG/CR-6620 (May 1999), "Testing of dc-Powered Actuators for Motor-Operated Valves."

On June 23, 2000, the BWROG forwarded Topical Report NEDC-32958 (March 2000), "BWR Owners' Group dc Motor Performance Methodology - Predicting Capability and Stroke Time in dc Motor-Operated Valves," to the NRC staff for information. On October 2, 2000, the BWROG recommended an implementation schedule of 12 months or the first refueling outage (whichever is later) for first priority MOVs (those with one- or two-cycle JOG static test frequencies), and two refueling outages for second priority MOVs (remaining GL 96-05 MOVs) with a start date of when the NRC acknowledged the methodology. On August 1, 2001, the NRC issued Regulatory Issue Summary (RIS) 2001-15, "Performance of dc-Powered Motor-Operated Valve Actuators," that informs licensees of the availability of improved industry guidance for predicting dc-powered MOV actuator performance. In RIS 2001-15, the NRC staff stated that, based on a sample review, the BWROG methodology represents a reasonable approach to improvement of past industry guidance for predicting dc-powered MOV stroke time and output. The staff considers the BWROG methodology to be applicable to Boiling Water Reactor (BWR) and Pressurized Water Reactor plants because of similarity in the design and application of dc-powered MOVs. With the availability of the new BWROG methodology, the staff considers that the regulatory issue of adequate prediction of dc-powered MOV performance can be effectively resolved through implementation of improved industry guidance. During a public meeting on March 4, 2004, the BWROG stated that all of its members had completed the implementation of the improved dc motor methodology for the first priority MOVs and that its members were in the process of implementing the methodology for the second priority MOVs. The BWROG did not report any significant concerns or problems with the implementation of the improved dc motor methodology.

Effects of Aging on MOV Internal Surfaces

In support of the NRC review of the JOG program, the NRC sponsored studies at INEEL and Battelle Memorial Institute in Columbus, Ohio, of the effects of aging on Stellite 6 which is used on sliding friction surfaces in valves. The tests of specimens in environments of temperature, pressure, and water chemistry typical of BWR nuclear plants were intended to determine the effects of film buildup on seating surfaces and the impact of the film on valve performance. The test results are provided in INEEL/EXT-99-00116 (April 1999), "Summary and Evaluation of NRC-Sponsored Stellite 6 Aging and Friction Tests," and NUREG/CR-6807 (March 2003), "Results of NRC-Sponsored Stellite 6 Aging and Friction Testing." The results of the aging tests identified the presence of a very thin oxide film after exposure times of only a few days. The test results indicated that friction increases as the test specimens age with the friction stabilizing prior to 120 days of aging. In general, the first test stroke revealed higher friction than succeeding strokes. The friction was reduced during subsequent strokes as the oxide film was removed. From the test program, periodic valve operation does not appear to have a significant effect on friction. However, valve operation shortly before a test might have an impact on the test results.

Effects of Aging and Temperature on MOV Stem Lubricants

To provide additional support for the NRC review of long-term MOV programs, the NRC sponsored a study at INEEL of the aging of stem lubricants and the effects of ambient temperature on their lubricating properties. The results of the research are provided in NUREG/CR-6750 (October 2001), "Performance of MOV Stem Lubricants at Elevated Temperature," and NUREG/CR-6806 (September 2002), "MOV Stem Lubricant Aging Research." The reports note that only a limited sample size was used in the test program. Nevertheless, the test results indicated that the stem friction coefficient for some lubricants can increase significantly under high ambient temperature conditions. The increased stem friction coefficient can cause a loss in the thrust delivered by the MOV motor actuator. For the valve stem tested, the program found that the new MOV Long Life lubricant performed similarly or in an improved manner to other lubricants previously tested.

Plant-Specific MOV Periodic Verification Program Review

Each U.S. nuclear power plant licensee submitted a description of plans for periodic verification of the design-basis capability of safety-related MOVs in response to GL 96-05. The NRC staff reviewed the licensee submittals and conducted inspections of GL 96-05 programs at a sample of nuclear plants. The staff prepared an SE to document its review of the response to GL 96-05 by each licensee. Where a licensee committed to implement the JOG program, the NRC staff relied to a significant extent on that commitment in preparing the SE without the need for plant-specific inspection activity in most instances. The NRC staff reviewed GL 96-05 programs of licensees that did not commit to the JOG program by a separate process of submittals and inspections, as appropriate. The NRC has completed its review of GL 96-05 programs for each active U.S. nuclear power plant. As licensees implement their long-term MOV programs including incorporation of the JOG program results, the NRC will monitor those programs using Inspection Procedure 62708, "Motor-Operated Valve Capability," as part of the NRC reactor oversight program.

Importance of MOV Followup Activities

The NRC staff continues to monitor plant-specific issues that could impact the capability of safety-related MOVs to perform their design-basis functions. For example, the NRC issued Information Notice (IN) 2003-15 (September 5, 2003), "Importance of Followup Activities in Resolving Maintenance Issues," to remind licensees that followup activities to verify implementation of corrective actions are an important part of a successful plan to resolve maintenance issues for safety-related components. In IN 2003-15, the NRC staff discussed the failure of an MOV at a U.S. nuclear power plant in January 2003 when its motor pinion gear moved along the motor shaft, and caused the motor to stall when contacting the declutch mechanism. In response to the MOV failure, the licensee inspected over 300 MOVs and found many deficiencies in motor pinion gear connections despite a long history of related industry information. When responding to operating experience and component performance information, it is important to have a clear plan of action to identify specific potentially affected components, and to address and track them to completion in a reasonable time based on their safety significance. The revision of maintenance procedures will only resolve a generic issue if the revised procedures are implemented during work activities. Where revised procedures are not implemented, the potential for common-cause failure can continue to exist for affected components in multiple plant systems.

IV. ASME ACTIVITIES TO IMPROVE MOV INSERVICE TESTING AND QUALIFICATION

The ASME Code specifies that stroke-time testing of MOVs be conducted as part of the IST programs of nuclear power plants on a quarterly frequency where practical. The NRC and the industry have long recognized the limitations of stroke-time testing as a means of assessing the operational readiness of MOVs to perform their design-basis safety functions. The NRC requires U.S. nuclear power plant licensees implementing the ASME OM Code to supplement the quarterly MOV stroke-time testing specified in the Code with a program to verify MOV design-basis capability on a periodic basis.

In response to concerns regarding the adequacy of MOV stroke-time testing, the ASME Operations and Maintenance Code Committee developed performance-based ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR Power Plants, OM Code 1995 Edition; Subsection ISTC." As an alternative to quarterly stroke-time testing, ASME Code Case OMN-1 allows periodic exercising of all safety-related MOVs once per refueling cycle and periodic diagnostic testing under static or dynamic conditions, as appropriate, on a frequency determined by MOV performance in terms of margin and degradation rate. In GL 96-05, the NRC staff noted that the method in ASME Code Case OMN-1 could be used as part of a licensee's response to the generic letter.

In Regulatory Guide (RG) 1.192 (June 2003), "Operation and Maintenance Code Case Acceptability, ASME OM Code," the NRC staff indicates that ASME Code Case OMN-1 is acceptable in lieu of stroke-time testing in the 1995 Edition up to and including the 2000 Addenda of the OM Code when applied with provisions for leakage rate testing. The NRC staff also indicates that licensees who implement Section XI of ASME BPV Code may use OMN-1 in lieu of stroke-time testing subject to the RG 1.192 conditions. The NRC staff states that licensees who implement OMN-1 must apply all of its provisions. The conditions for use of OMN-1 in RG 1.192 are:

(1) The adequacy of diagnostic test interval for each MOV must be evaluated and adjusted not later than 5 years or 3 refueling outages (whichever is longer) from OMN-1 implementation.

(2) If the exercise intervals for high-risk MOVs are extended, licensees must ensure that the increase in Core Damage Frequency and risk is small and consistent with the Commission's Safety Goal Policy Statement.

(3) Licensees must categorize MOVs using the methodology in ASME Code Case OMN-3 consistent with the RG 1.192 conditions, or use other MOV risk-ranking methodologies accepted by NRC with the conditions in the applicable safety evaluations.

The NRC staff also notes in RG 1.192 that the benefits of performing a particular test should be balanced against the potential adverse effects.

In RG 1.192, the NRC staff indicates that Code Case OMN-11, "Risk-Informed Testing for Motor-Operated Valves," is acceptable in supplementing the risk insights in Paragraph 3.7 of OMN-1 with the following conditions:

(1) In addition to the IST provisions of Paragraph 3 of OMN-11, MOVs within the scope of OMN-1 that are categorized as Low Safety Significant Components (LSSCs) must satisfy the other provisions of OMN-1, including the determination of proper MOV test intervals.

(2) Paragraph 3(a) of OMN-11 must be interpreted as allowing the provisions of Paragraph 3.5 of OMN-1 related to similarity and test sample to be relaxed when grouping LSSC MOVs. Provisions in Paragraph 3.5 related to evaluation of test results, sequential testing, and analysis of test results per Paragraph 6 of OMN-1 continue to be applicable to all MOVs within the OMN-1 scope.

(3) If extending high-risk MOV exercise intervals, licensees must ensure that the increase in Core Damage Frequency and risk is small and consistent with the Commission's Safety Goal Policy Statement.

In RG 1.192, the NRC staff also notes that the condition regarding allowable methodologies for MOV risk ranking also applies to OMN-11.

The NRC staff has granted requests from nuclear power plant licensees to apply OMN-1 as an alternative to the quarterly MOV stroke-time testing in their particular ASME Code of record. Currently, ASME is preparing a revision to OMN-1 to improve its application to more nuclear power plants by clarifying several aspects of the code case while retaining the safety improvement that is achieved through increased knowledge of the design-basis capability of MOVs obtained from diagnostic testing. Over the longer term, it is recommended that ASME replace the quarterly MOV stroke-time testing specified in the ASME Code with performance-based provisions similar to those in ASME Code Case OMN-1.

With respect to MOV qualification, the Subcommittee on Qualification of Valve Assemblies (SC-QV) of the ASME Committee on Qualification of Mechanical Equipment used in Nuclear Facilities has prepared a proposed revision to Section QV, "Functional Qualification Requirements for Active Valve Assemblies for Nuclear Power Plants," of the ASME Standard QME-1, "Qualification of Active Mechanical Equipment used in Nuclear Power Plants." The proposed revision to Section QV to QME-1 reflects valve performance information obtained from nuclear industry programs and NRC-sponsored research since development of the QME-1 standard in the 1980s. At a meeting on February 23, 2004, SC-QV completed its resolution of comments on the proposed revision to Section QV, and planned to forward the proposed revision to Section QV to the QME main committee for balloting.

V. PRESSURE LOCKING AND THERMAL BINDING OF GATE VALVES

One typical method that "pressure locking" can occur in flexible-wedge and double-disk gate valves is when pressure in the bonnet is higher than the line pressure on both sides of a closed disk and the valve actuator is not capable of overcoming the additional thrust required as a result of the differential pressure. Thermal binding is generally associated with a solid- or flexible-wedge gate valve that is closed at high temperature and is allowed to cool before reopening is attempted such that mechanical interference occurs because of contraction of the valve body on the disk wedge. On August 17, 1995, the NRC issued GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," to request that licensees perform, or confirm that they had previously performed, (1) evaluations of the operational configurations of safety-related, power-operated (including motor-, air-, and hydraulically operated) gate valves for susceptibility to pressure locking and thermal binding; and (2) further analyses, and any needed corrective actions, to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing their safety functions within the current licensing basis of the facility.

NUREG/CR-6611 (May 1998), "Results of Pressure Locking and Thermal Binding Tests of Gate Valves," describes testing sponsored by the NRC Office of Nuclear Regulatory Research at INEEL to study pressure locking and thermal binding of gate valves in support of GL 95-07.

The NRC staff has completed its review of licensee responses to GL 95-07 through issuance of an SE addressing each active U.S. nuclear power plant.

VI. CONCLUSIONS

As a result of problems identified in the 1980s with MOV performance at nuclear power plants, the NRC issued GLs 89-10 and 96-05 requesting that licensees verify initially and periodically the design-basis capability of MOVs in safety-related systems at nuclear power plants. In response to GL 96-05, the nuclear power plant owners groups developed an industry-wide JOG program for periodic verification of the design-basis capability of safety-related MOVs. The NRC accepted the JOG program as an industry-wide response to GL 96-05 with respect to age-related valve degradation. The NRC issued GL 95-07 requesting that licensees ensure that safety-related power-operated gate valves susceptible to pressure locking or thermal binding are capable of performing their safety functions. Licensees of all active U.S. operating reactor units have completed their programs to verify initially the design-basis capability of safety-related MOVs in response to GL 89-10, and to address potential pressure locking and thermal binding of safety-related power-operated valves in response to GL 95-07. Licensees are currently implementing their long-term MOV programs in response to GL 96-05. The NRC staff has completed its review of GL 96-05 programs established at individual nuclear plants through significant reliance on licensee commitments to implement the JOG program on MOV periodic verification. The NRC staff is reviewing the JOG final topical report that describes the long-term periodic verification of the design-basis capability of MOVs for use by licensees as part of their commitments to GL 96-05. In its regulations, the NRC has directed licensees implementing the ASME OM Code to supplement the quarterly MOV stroke-time testing in their IST programs with a program to periodically verify MOV design-basis capability. The NRC staff has granted requests from licensees to apply performance-based ASME Code Case OMN-1 as an alternative to the quarterly MOV stroke-time testing in their ASME Code of record. The NRC has accepted generic use of ASME Code Case OMN-1 as an alternative to MOV stroke-time testing in RG 1.192. The NRC continues to monitor licensee activities related to the performance of safety-related MOVs through the reactor oversight program.

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RULEMAKING ACTIVITIES ON INSERVICE TESTING

Stephen G. Tingen
Mechanical and Civil Engineering Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Abstract

Section 50.55a of Title 10 of the Code of Federal Regulations (10 CFR 50.55a) establishes requirements for the application of codes and standards in the performance of inservice testing of components used in nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC) periodically updates 10 CFR 50.55a to incorporate by reference recent editions and addenda to the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Plants (OM Code) for inservice testing (IST) of pumps and valves used in nuclear power plants. The NRC is currently updating 10 CFR 50.55a to incorporate by reference the 2001 Edition through 2003 Addenda of the ASME OM Code. This proposed action will accord the provisions in the 2001 Edition and the 2002 and 2003 Addenda to the ASME OM Code the same legal status as the earlier editions and addenda of the ASME OM Code that have been incorporated by reference in 10 CFR 50.55a. This paper will present the status of this rulemaking and other rulemakings that are related to inservice testing of pumps and valves.

I. Incorporation By Reference of a Later Edition and Addenda of ASME Code

In Commission paper SECY-03-0078 (May 15, 2003), the NRC staff requested approval of the Commission for the initiation of a rulemaking to amend 10 CFR 50.55a to incorporate by reference the following: (1) the 2001 Edition, 2002 Addenda, and 2003 Addenda of Division 1 rules of Section III, “Rules for Construction of Nuclear Power Plant Components,” of the ASME Boiler and Pressure Vessel Code (BPV Code); (2) the 2001 Edition, 2002 Addenda, and 2003 Addenda of Division 1 rules of Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” of the ASME BPV Code; and (3) the 2001 Edition, 2002 Addenda, and 2003 Addenda of the ASME OM Code. To improve the timeliness of NRC review and approval of new editions and addenda of the ASME Code, the staff proposed in SECY-03-0078 to conduct rulemakings to keep current the ASME Code editions and addenda incorporated by reference in 10 CFR 50.55a at approximately 2 to 3 year intervals. The Commission approved the staff’s proposal in a staff requirements memorandum dated May 30, 2003.

On January 7, 2004 (69 FR 879), the NRC published a proposed rule in the Federal Register that presented an amendment to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” that would revise the requirements for construction, inservice inspection (ISI), and IST of nuclear power plant components. The proposed revision to § 50.55a(b)(3) would incorporate by reference the 2001 Edition and the 2002 and 2003 Addenda of the ASME OM Code.

The proposed amendment would revise the existing modifications and limitations for quality assurance, motor-operated valve testing, Subsection ISTD on snubbers, and exercise interval for manual valves in §§ 50.55a(b)(3)(i), 50.55a(b)(3)(ii), 50.55a(b)(3)(v), and 50.55a(b)(3)(vi), respectively, to apply to the 2001 Edition through 2003 Addenda of the ASME OM Code. The modifications and limitations in §§ 50.55a(b)(3)(i), 50.55a(b)(3)(ii), 50.55a(b)(3)(v), and 50.55a(b)(3)(vi) would continue to apply to the 2001 Edition through 2003 Addenda of ASME OM Code because the earlier Code provisions on which these regulations were based were not revised in the 2001 through 2003 Addenda of the ASME OM Code to resolve the underlying issues which led the NRC to impose the modifications and limitations on the ASME Code provisions.

The proposed amendment would revise the existing quality assurance requirements in § 50.55a(b)(3)(i) to state that paragraph ISTA-1500 of Subsection ISTA in the ASME OM Code is applicable when using the 1998 Edition and

This paper was prepared by staff of the U.S. Nuclear Regulatory Commission. It may present information that does not currently represent an agreed-upon NRC staff position. NRC has neither approved nor disapproved the technical content.

later editions and addenda of the Code. Subsections of the ASME OM Code were renumbered in the 1998 Edition; therefore, § 50.55a(b)(3)(i) would be revised to account for the renumbering. The proposed revision does not change IST requirements in a substantive manner.

The proposed amendment would revise § 50.55a(b)(3)(iii) to eliminate the authorization in this paragraph to use Code Case OMN-1. Code Case OMN-1 is now authorized by Regulatory Guide 1.192, Operation and Maintenance Code Case Acceptability, ASME OM Code. Regulatory Guide 1.192 was incorporated by reference into § 50.55a in a final rule dated July 8, 2003 (68 FR 40469). Thus, it is no longer necessary to authorize the use of Code Case OMN-1 in § 50.55a(b)(3)(iii) because this code case is now included in Regulatory Guide 1.192.

The proposed amendment would revise the existing modification for the check valve monitoring program in § 50.55a(b)(3)(iv) to limit its application to the 1995 Edition through 2002 Addenda of the ASME OM Code. The modification in § 50.55a(b)(3)(iv) would not apply to the 2003 Addenda of the ASME OM Code because the earlier Code provisions on which this regulation was based were revised in the 2003 Addenda of the ASME OM Code to resolve the underlying issues which led the NRC to impose the modification to the ASME Code provisions. The check valve monitoring program requirements in Appendix II of the 2003 Addenda of the ASME OM Code are equivalent to the check valve monitoring program requirements in § 50.55a(b)(3)(iv).

Public Meetings

On August 25, 2003, NRC staff from the NRC Office of Nuclear Reactor Regulation held a public meeting in Scottsdale, Arizona. The purpose of the public meeting was to present, and obtain stakeholder feedback on, the proposed rulemaking to amend 10 CFR 50.55a to incorporate by reference the 2001 Edition through 2003 Addenda of Sections III and XI, Division 1, of the ASME BPV Code. These two sections of the Code provide requirements for the design and ISI of nuclear power plant components.

The NRC staff presented its issues associated with the use of 2001 Edition and 2002 and 2003 Addenda of Sections III and XI of the ASME BPV Code. The public meeting was held in the evening at the same location that ASME Sections III and XI committees were meeting to enhance stakeholder participation. Approximately 60 members of the public attended the meeting. Most of the public that attended the meeting were members of the ASME. There was a good exchange of information between the NRC staff and the public during the meeting. The staff noted, however, that the verbal feedback does not preclude the need to submit written comments when the proposed rule is issued. Members of the ASME commented on several of the issues and provided additional information for the NRC staff to consider. The NRC staff evaluated the additional information provided at the meeting and revised sections of the proposed rule based on comments received during the meeting.

On February 23, 2004, the NRC held a public meeting in St. Petersburg, Florida, to discuss NRC's proposed rule (69 FR 879) to incorporate by reference into 10 CFR 50.55a the 2001 Edition (up to and including the 2003 Addenda) of ASME Boiler and Pressure Vessel Code, Section III. Specifically, the public was invited to comment on those portions of the latest Code related to changes in the seismic design rules for piping systems. The public meeting was held in conjunction with the ASME Code committee meetings that week. Approximately 40 persons attended the public meeting. The latest changes to the Code rules represented a culmination of effort in place since 1995 when the NRC placed a restriction in 10 CFR 50.55a on the use of the revised ASME Code rules for piping seismic design that first appeared in the 1994 Addenda. In 1995, the ASME Code assigned a special task group to resolve the NRC's concerns, and the task group's effort resulted in the revised Code rules published in the 2001 Edition up to and including the 2003 Addenda. In the proposed rule, the NRC staff would accept the new ASME Code piping seismic rules with six modifications and limitations. At the public meeting, the NRC staff heard presentations by three ASME piping experts including a Japanese seismic team involved in dynamic testing of piping systems.

The NRC plans to continue to conduct meetings to obtain stakeholder feedback on future proposed rulemakings that amend 10 CFR 50.55a to incorporate by reference a later edition and addenda of the ASME Code when significant issues with the use of the later edition and addenda of the ASME BPV or OM Code are identified. The NRC staff did not identify any significant issues with the use of the 2001 Edition through 2003 Addenda of the ASME OM Code; therefore, the NRC staff did not consider a public meeting to be necessary.

II. Incorporation By Reference of "Code Case" Regulatory Guides

The ASME develops and publishes the BPV Code, which contains the Code requirements for design, construction, and ISI of nuclear power plant components, and the OM Code, which contains Code requirements for IST of nuclear power plant components. In response to Code user requests, the ASME develops Code cases for the BPV and OM Code which provide alternatives to the Code requirements under special circumstances.

The NRC staff reviews ASME Code Cases, determines their acceptability, and publishes its findings in NRC Regulatory Guides (RGs). The RGs are revised periodically as new Code cases are published by the ASME. On July 8, 2003, the NRC issued a final rule (68 FR 40469) which initiated the practice of incorporating by reference the RGs listing the acceptable and conditionally acceptable ASME Code cases in § 50.55a. Thus, NRC RG 1.84 (Revision 32), Design, Fabrication, and Materials Code Case Acceptability, ASME Section III; RG 1.147 (Revisions 0 through 13), Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1; and RG 1.192, Operation and Maintenance Code Case Acceptability, ASME Code, were incorporated into the NRC's regulations. The NRC is now proposing to incorporate by reference RG 1.84 (Revision 33) and RG 1.147 (Revision 14) to replace earlier revisions of these RGs in the NRC's regulations.

The NRC staff reviewed Code Cases OMN-1 through OMN-13 for inclusion into the version of RG 1.192 that is currently incorporated by reference in 10 CFR 50.55a. The NRC staff is not proposing a revision to RG 1.192 at this time because additional code cases have not been published by the ASME OM Code.

III. Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors

The proposed rule dated May 16, 2003 (68 FR 265110) would amend NRC regulations to provide an alternative approach for establishing the requirements for treatment of structures, systems, and components (SSCs) for nuclear power reactors using a risk-informed method of categorizing SSCs according to their safety significance. The proposed amendment would revise requirements with respect to "special treatment," that is, those requirements that provide increased assurance (beyond normal industrial practices) that SSCs perform their design basis functions. This proposed amendment is further discussed in the risk-informed IST session of this symposium.

IV. Conclusion

The final rule to update 10 CFR 50.55a to incorporate by reference the 2001 Edition through 2003 Addenda of the ASME OM Code is scheduled to be published in the Federal Register in October 2004. The final rule will become effective 30 days from date of publication in the Federal Register. Licensees of nuclear power plants would be required to use the 2001 Edition and the 2002 and 2003 Addenda of the ASME OM Code when updating IST programs in subsequent 120-month inspection intervals under § 50.55a(f)(4)(ii). The proposed rule to amend 10 CFR 50.55a to incorporate by reference the NRC's RGs that address the use of Code Cases prepared by the ASME BPV and OM Code will not include RG 1.192. The final rule that would add 10 CFR 50.69, "Risk-Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," is scheduled to be completed in mid-2004.

NUCLEAR POWER PLANT PUMP AND VALVE INSERVICE TESTING ISSUES

David Terao
Gurjendra S. Bedi
Mechanical and Civil Engineering Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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Abstract

This paper summarizes a number of pump and valve inservice testing issues raised since the Seventh Nuclear Regulatory Commission (NRC)/American Society of Mechanical Engineers (ASME) Symposium on Valve and Pump Testing. The issues have generic applicability to United States nuclear power plants. Among the issues addressed are the comprehensive pump test (CPT), frequency response range of vibration measuring transducers, and online testing of check valves.

INTRODUCTION

The NRC staff has encountered a number of pump and valve inservice testing (IST) issues since the Seventh NRC/ASME Symposium on Valve and Pump Testing in 2002. This paper discusses pump issues involving the comprehensive pump test (CPT) and the frequency range of vibration-measuring transducers and valve issues involving online testing of check valves. The paper discusses the relief requests received related to these issues and the NRC safety evaluations of the requests. Some current staff positions and actions in these areas are discussed.

COMPREHENSIVE PUMP TEST ISSUES

On September 22, 1999, the staff's endorsement of the 1995 Edition of the ASME Operation and Maintenance (OM) Code up to and including the 1996 Addenda was published in the Federal Register (Vol. 64, No. 183). With this rulemaking came revised requirements for IST. The 1995 ASME OM Code includes a new set of pump testing requirements which are collectively known as the "comprehensive pump test." The CPT allows less rigorous pump testing to be performed for certain pumps on a quarterly frequency while requiring a pump test to be performed with more accurate flow instrumentation every 2 years at ±20 percent of pump design flow. The CPT was developed with the knowledge that some pumps, such as containment spray pumps, cannot be tested at the required high flow rates because of limitations of system design. All ASME OM Code editions and addenda, issued since 1995 contain CPT requirements.

Licensees have started to update their IST programs, as required by 10 CFR 50.55a, to the 1995 Edition through the 1996 Addenda of the ASME OM Code. Relief requests have been submitted to the NRC staff to propose alternative testing to the CPT pump design flow requirements because the requirements for certain pumps have been determined by the licensee to be either a burden or impractical. This paper only summarizes various issues related to the CPT in these proposed relief requests and the NRC staff's published evaluation. The intent of this paper is to summarize the current evaluations and present licensees with issues to consider if they are contemplating similar licensing actions.

OM Code Subsection ISTB-1995 introduces a new approach to pump testing by dividing pumps into two basic groups. The pump grouping criteria of ISTB are based on the way the pumps are operated at the plant. There are two groups: normally or routinely operated pumps (group A) and standby pumps (group B). The Code identifies four type of tests: preservice test, Group A test, Group B test, and comprehensive test. All pumps receive a preservice test followed quarterly by the test associated with the pump category (Group A test for Group A pump, etc.). A comprehensive test may be substituted for a Group A test or

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Group B test. A Group A test may be substituted for a Group B test. A preservice test may be substituted for any inservice test.

As a point of information, the OM-6 pump testing standard was issued in October 1990 as OM Code-1990, Subsection ISTB (ASME, 1990). The CPT change was written against the 1990 Subsection ISTB. The 1995 OM Code, ISTB 4.3(e)(1), requires that reference values be established within ±20% of design flow for the comprehensive test.

The staff authorized or denied alternatives proposed in the following relief requests, as documented in their NRC safety evaluation:

- Seabrook Station, Unit 1
- North Anna Power Station, Units 1 and 2
- Calvert Cliffs Nuclear Power Plant, Units 1 and 2
- H. B. Robinson Steam Electric Plant, Unit 2
- Vermont Yankee
- Sequoyah, Units 1 and 2
- Monticello Nuclear Generating Plant

This paper only summarizes the various relief requests and the safety evaluation results. Licensees can review the details of a particular relief request safety evaluation in the publicly available NRC Agency wide Documents Access and Management System (ADAMS). The NRC ADAMS number associated with the relief request is shown in the Remarks column of the attached summary table.

Seabrook Station, Unit 1

The licensee of Seabrook Station submitted relief request PR-1 on March 21, 2000. The proposed alternative to the Code reference value requirements of ISTB 4.3.e(1) for the containment spray pumps CBS-P-9A and CBS-P-9B was authorized pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that the alternative provided an acceptable level of quality and safety for an interim period of 2 years.

During the interim period, the licensee was requested to reevaluate the current testing to assess the ability to detect degradation as was intended by the OM Code-1995 pump test strategy. The NRC safety evaluation stated: "This may entail more detailed analysis of the IST data, consultation with the manufacturer, or running additional tests as appropriate. If the licensee cannot further demonstrate that the proposed testing is an acceptable alternative, then appropriate compensatory actions should be proposed to supplement the alternative testing. Possible strategies or combinations of strategies include: 1) testing at the best efficiency point (BEP) on a much longer interval; 2) commitment to perform additional performance monitoring on the containment spray pumps; 3) adjustment of acceptance criteria; and/or 4) continuation of the current Code testing, including taking overall vibration data quarterly."

The licensee resubmitted a revised relief request PR-1 on October 28, 2002, with additional information and proposed compensatory actions. The NRC staff concluded that meeting the requirements of ISTB 4.3.e(1) for the containment spray pumps CBS-P-9A and CBS-P-9B was impractical at that time. The staff also concluded that testing the containment spray pumps at 63 percent of the pump best efficiency point using the recirculation flow lines, together with the proposed compensatory actions, provided reasonable assurance of the operational readiness of the containment spray pumps.

Based on a review of the information provided by the licensee, the NRC staff granted the licensee's request for relief and the proposed alternatives to the Code requirements of ISTB 4.3.e(1) for the containment spray pumps pursuant to 10 CFR 50.55a(f)((6)(i) on the basis that the Code requirements were impractical.

North Anna Power Station, Units 1 and 2

The licensee of North Anna Station submitted relief request P-6 on June 4, 2001. Based on a review of the information provided by the licensee, the NRC staff concluded that the licensee's proposed alternative to the Code-required number of data points on pump test curves and to the reference value requirements of Table ISTB 4.1(a) and paragraph ISTB 4.3(e) for recirculation spray pumps 1-RS-P-2A and 2B, and 2-RS-P-2A and 2B was authorized pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis that compliance with the specified requirement resulted in a hardship without a compensating increase in the level of quality and safety. The NRC staff further concluded that the alternative provided reasonable assurance of the operational readiness of the pump.

The licensee committed to include all the outside recirculating pumps in the North Anna Predictive Maintenance Program. Under this program, if the measured parameters are outside the normal operating range or are determined by the analysis to be trending towards an unacceptable degraded state, the licensee will take appropriate actions, including monitoring additional parameters, reviewing component-specific information to identify the cause, and removing the pump from service to perform maintenance.

Calvert Cliffs Nuclear Power Plant, Units 1 and 2

The licensee of Calvert Cliffs Nuclear Power Plant submitted relief request PR-12 on January 4, 2002. The NRC staff concluded that the use of the OM Code, Subsection ISTB, 1995 Edition with 1996 Addenda (instead of OM-6, 1987/88) for the pump testing for the Calvert Cliffs Nuclear Power Plant was acceptable and approved the request pursuant to 10 CFR 50.55a(f)(4)(iv). This relief request was for only for high-pressure coolant injection (HPCI), low- pressure coolant injection (LPCI), and the containment spray pumps.

H. B. Robinson Steam Electric, Plant Unit 2

The licensee of H. B. Robinson submitted relief request IST-RR-3 on August 24, 2001. The licensee proposed an alternative to perform a reduced-flow comprehensive test for containment spray pumps A and B in lieu of a full-flow comprehensive test as required by OM Code, paragraph ISTB 4.3(e)(1). This relief was authorized pursuant to 10 CFR 50.55a(f)(6), for an interim period of 2 years on the basis that the Code-required test was impractical to perform without significant plant modification, that the interim alternative otherwise met the criteria of 10 CFR 50.55a(f)(6)(i), and that the interim relief would allow time for the licensee to explore other alternatives, make necessary plant modifications for performing the required test, or submit a revised relief request.

The licensee of H. B. Robinson submitted revised relief request IST-RR-3 on April 15, 2003, with additional information. The NRC staff concluded that the licensee's proposed alternative did not provide an acceptable level of quality and safety and did not explain why compliance with Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety; therefore, the request was denied. The licensee subsequently modified the system's design to install a full-flow test line to allow comprehensive pump testing in accordance with the Code requirements.

Vermont Yankee

The licensee of Vermont Yankee submitted relief request RR-P01 on January 22, 2003, for the service water (SW) pumps. The staff concluded that compliance with the Code-required Group A quarterly flow test (and associated differential pressure testing) of the SW pumps would require significant redesign of the SW system. Relief was granted pursuant to 10 CFR 50.55a(f)(6)(i) on the basis of the impracticality of performing inservice testing in accordance with ASME OM Code requirements. The Code-specified comprehensive pump test shall be performed on the SW pumps on a refueling outage frequency (every 18 months). Vibration measurements, including full spectral analyses, will be performed quarterly with vibration measurements assessed in accordance with the Code (using quarterly differential pressure measurements to establish a variable reference value). The licensee's proposed alternative testing and analyses provided reasonable assurance of the pumps' operational readiness.

Sequoyah, Units 1 and 2

The licensee of Sequoyah, Units 1 and 2, submitted relief requests RP-09 and RP-10 on April 17, 2002, for the turbine driven auxiliary feedwater (TDAFW) pumps. On the basis that the NRC incorporated by reference in 10 CFR 50.55a(b) the 1995 Edition through 1996 Addenda of the OM Code, the use of the OM Code, Subsection ISTB, 1995 Edition with 1996 Addenda, for the CPT for the Sequoyah, Units 1 and 2, TDAFW pumps was approved pursuant to 10 CFR 50.55a(f)(4)(iv).

Monticello Nuclear Generating Plant

The licensee of Monticello submitted relief request PR-06 on May 6, 2003, for the HPCI pumps. The NRC staff concluded that the licensee's methodology to establish and use reference curves in the performance of the Group B and comprehensive tests of HPCI pump P-209 at Monticello Nuclear Generating Plant (MNGP) provided an acceptable level of quality and safety because MNGP used the method approved by the NRC staff in Code Case OMN-9 to establish a reference value curve for pump differential pressure and flow rate, and the requirement for conducting pump IST within ±20 percent of the pump design flow rate was not affected.

The NRC staff concluded that Monticello's request to use reference curves as part of an alternative testing methodology to satisfy the provisions in paragraphs ISTB 5.2.2(a) and ISTB 5.2.3(a) of the ASME OM Code for the Group B and comprehensive tests, respectively, of the HPCI pump provided an acceptable level of quality and safety. On this basis, the NRC staff authorized Monticello's proposed alternative in accordance with 10 CFR 50.55a(a)(3)(i).

Summary of Nuclear Power Plants Requesting Relief related to Comprehensive Pump Test

Plant	Relief Request	OM Code Section (year)	Comprehensive Test		Reason	Remark
			Flow (% of design)	TDH-Flow (total dynamic head flow) curve		
Seabrook, Unit 1 (50-443) Cont. spray pumps	PR-1, March 21, 2000 (2nd 10-year IST)	ISTB 4.3.e.1 1995/96	1900 gpm (63%)	Approx. Flat	The licensee proposed only recirculation line flow for comprehensive test.	Authorized for 2 years, interim ML003760787
	Revised PR-1, October 28, 2002	ISTB 4.3.e.1 1995/96	1900 gpm (63%)	Approx. Flat	(1) Repeatable results (2) Spray pumps are included in Predictive Maintenance Monitoring Equipment Program, which will enhance vibration monitoring and analysis and periodic lube oil analysis. (3) Corrective action required, if unacceptable degraded condition found	Relief granted, 10 CFR 50.55a(f)(6)(i) ML031070510
Calvert Cliffs, Units 1 and 2 (50-317) (50-318) HPCI, LPCI, spray pumps	PR-12 Jan 4, 2002 (3nd 10-year IST)	Use of ISTB 1995/96 instead of OM-6 1987/88	N/A	N/A	The licensee proposed to use OM Code 1995 through 1996 Addenda, Subsection ISTB, which allows to use of comprehensive pump test.	Approved pursuant to 10 CFR 50.55a(f)(4)(iv) ML021000690
North Anna, Units 1 & 2 (50-338) (50-339) Spray Pumps	P-6, June 4, 2001 (3nd 10-year IST)	ISTB 4.3.e.1 1995/96	1500 gpm (40%)	Flow varies with TDH	(1) Full-flow test was performed in Unit 2 during construction phase. (2) Flow increases with lower TDH. (3) Spray pumps are included in Predictive Maintenance Program for additional testing, trending, and diagnostic analysis. (3) Corrective action required if unacceptable degraded condition found. (4) More restrictive differential pressure requirement than Table ISTB 5.2.3-1 to ensure pumps can deliver required accident flow.	Authorized 10 CFR 50.55a(3)(ii) ML020280439

Summary of Nuclear Power Plants Requesting Relief related to Comprehensive Pump Test

Plant	Relief Request	OM Code Section (year)	Comprehensive Test		Reason	Remark
			Flow (% of design)	TDH-Flow (total dynamic head flow) curve		
Sequoyah, Units 1 & 2 (50-327) (50-328) AFW (aux. feedwater) pumps	RP-09 & RP-10, April 17, 2002 (2nd 10-year IST)	Use of ISTB 1995/96 instead of OM-6 1987/88	N/A	N/A	The licensee proposed to use OM Code 1995 through 1996 Addenda, Subsection ISTB, which allows to use of comprehensive pump test.	Approved pursuant to 10 CFR 50.55a(f)(4)(iv) ML021970279
H.B. Robinson, Unit 2 (50-261) Spray pumps	IST-RR-3 August 24, 2001 (4th 10-year IST)	ISTB 4.3.e.1 1995/96	240 gpm (20%)	N/A	The licensee proposed only recirculation line flow (20% of maximum flow) for comprehensive pump test. Pump was never tested at design flow in installed condition.	Authorized for 2 years, interim ML040700790
	Revised IST-RR-3 April 15, 2003	ISTB 4.3.e.1 1995/96	Test at 33%	N/A	The licensee proposed (1) test at 33% maximum flow (2) vibration monitoring quarterly (3) $220,000 required for design modification to meet the code.	Relief denied. ML031780245
Vermont Yankee (50-271) Service water pumps	RR-P01 Jan 22, 2003 (4th 10-year IST)	ISTB3400 and Table ISTB-3400-1 OM-1998 2000 Add.	Full design flow	N/A	Service water pumps are Group A pumps. Licensee will perform CPT every refueling outage at full flow, but cannot perform Group A flow test quarterly due to existing SW system limitation. The licensee will measure vibration and differential pressure every quarter, plot flow-head curve quarterly to provide additional information, and perform lube analysis.	Relief granted due to impracticality per 10 CFR 50.55a(f)(6)(i) ML032020388
Monticello (50-263) HPCI pumps	PR-06 May6, 2003 (4th 10-year IST)	ISTB 5.2.2(a) and 5.2.3(a) OM-1995 1996 Add.	For details see relief request and OMN-9 guidelines	For details see relief request and OMN-9 guidelines	The licensee stated that use of an accurate reference value is very important for accurate trending and analysis. The licensee stated that the complexities of the flow control system for the HPCI pump make it difficult to exactly duplicate the reference points. The licensee proposed an alternative test method for the Group B and comprehensive tests of the HPCI pump. The alternative testing used the methodology of the Code Case OMN-9, "Use of a Pump Curve for Testing."	Authorized 10 CFR 50.55a(3)(ii) ML032060580

PUMP'S VIBRATION MEASURING INSTRUMENTS (TRANSDUCERS) ISSUE

The NRC has received relief requests from various licensees for relief from the provisions of ISTB 4.7.1(f) of the ASME OM Code for pumps with low pump shaft rotational speeds. Paragraph ISTB 4.7.1(f), "Frequency Response Range," requires that the frequency response range of the vibration-measuring transducers and their readout system shall be from one-third minimum pump shaft rotational speed to at least 1000 hertz (Hz).

Most of the licensees stated that procurement and calibration of instruments to cover the lower end of the Code-specified range was impractical due to the limited number of vendors supplying such equipment, the level of equipment sophistication required, and the equipment cost. Therefore, past relief requests were typically authorized pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis that compliance with the specified Code requirement would result in hardship without a compensating increase in the level of quality and safety. The NRC provided detailed safety evaluations authorizing these relief requests.

The NRC has learned that, due to technology advancement and research work performed in the field of instrumentation, vibration-measuring transducers meeting the Code requirements can be easily procured from various suppliers at a reasonably low cost.

Therefore, licensees are requested to carefully examine the availability, procurement, and related cost of the Code-required instruments (vibration-measuring transducers) before submitting a relief request to the NRC.

Recently, a similar relief request was received from the licensee of the Pilgrim Nuclear Power Station. After review, requests for additional information, and followup discussion by the NRC, the licensee withdrew the relief request and decided to install a new transducer, that met the Code requirements.

ONLINE TESTING OF CHECK VALVES ISSUES

In an effort to shorten refueling outages, many licensees are performing as much maintenance and testing, and as many other surveillance activities, as possible with the nuclear power plant online. For example, several licensees have submitted relief requests to the NRC to conduct inservice testing once per refueling cycle, rather than during a refueling outage as prescribed by the Code. Several factors should be taken into consideration in preparing (and evaluating) such relief requests to ensure that the proposed alternative provides an acceptable level of quality and safety.

If a licensee is testing a particular valve during refueling outages, it may be because the licensee determined that it was impractical to test the valve quarterly during operation or during cold shutdown. The inservice testing program should document the basis for deferring the testing from quarterly (and during cold shutdown) to refueling outages. Relief requests to perform testing with the nuclear plant online should be prepared in light of the refueling outage justification for each valve or group of valves affected. If necessary, the refueling outage justification should be revised to be consistent with the relief request.

Consideration should be given to whether the testing can be readily accomplished within the allowed outage time permitted by any applicable technical specification. In general, the time necessary to complete the testing should be significantly less than the allowed outage time. This general consideration is intended to avoid technical specification violations or the need to issue exigent technical specification amendments or notices of enforcement discretion.

Sometimes there is a tradeoff between testing these valves at power and testing them during outages (e.g., when there may be greater reliance on shutdown cooling or when other necessary equipment is out of service). Licensees should provide a risk-informed justification, either quantitative or qualitative, for why testing online is appropriate instead of testing during the refueling outage. Licensees should identify any compensatory measures to be established as a risk management action to reduce the risk impact of testing with the nuclear power plant at power. If relevant, licensees should provide information on how testing at power versus testing during refueling outages will affect scheduled maintenance work windows for the applicable system. Can this testing be done within these work windows or does this testing extend either the shutdown or at-power work windows? In calculating the difference in risk between testing at power and testing during refueling outages, a new estimate of the maintenance unavailabilities may need to be developed that will reflect the increased maintenance activities at power and the basis for the estimate should be documented.

At times, testing (or the disassembly and inspection of valves) during refueling outages can be more advantageous from a worker safety perspective when, for example, the system is cold and depressurized. Licensees should consider worker safety and discuss whether the valve or valves can be adequately isolated (e.g., leakage) when requesting that testing be performed with the nuclear plant online.

Several licensees have submitted relief requests to the NRC to take credit for maintenance activities performed to meet the requirements of 10 CFR 50.65 for inservice testing of components. The inservice testing requirements of 10 CFR 50.55a and the maintenance rule requirements of the 10 CFR 50.65 rules are two separate activities. Therefore, inservice testing activities and maintenance activities as required by 10 CFR Part 50 cannot be interchanged. The staff requests licensees not to submit relief requests to interchange maintenance rules activities with the inservice testing requirements.

CONCLUSION

The purpose of this paper is to make licensees aware of a number of pump and valve issues that the staff has encountered since the Seventh NRC/ASME Symposium on Valve and Pump Testing in 2002. Licensees who believe that some of the items discussed are applicable to their facilities may wish to review their current IST program and modify their program as appropriate.

REFERENCES:

NUREG Reports

NUREG/CP-0152, Vol. 4, “Proceedings of the Seventh NRC/ASME Symposium on Valve and Pump Testing,” July 2002.

NUREG/CR-6396, “Examples, Clarifications, and Guidance on Preparing Requests for Relief From Pump and Valve Inservice Testing Requirements.”

NUREG-1482, “Guidelines for Inservice Testing at Nuclear Power Plants.”

Generic Letters

GL 89-04, “Guidance on Developing Acceptable Inservice Testing Programs.”

Correspondence

March 21, 2000, letter from T. C. Feigenbaum of North Atlantic Energy Service Corporation to NRC, “Safety Evaluation of Relief Requests for the Second 10-year Interval Inservice Test Program Plan, Seabrook Station, Unit No. 1” (TAC No. A8532).

October 28, 2002, letter from J. M. Vargas, North Atlantic Energy Service Corporation, to Nuclear Regulatory Commission, “Seabrook Station, Inservice Testing Program for Pumps and Valves for Second 10-year Interval, Revision to Relief Request PR-1” (TAC No. MB6676).

June 4, 2001, letter from Virginia Electric and Power Company (Dominion) to NRC, “Virginia Electric and Power Company, North Anna Power Station Units 1 and 2 Inservice Testing Program for Pumps and Valves for Third 10-year Interval” (TAC Nos. MB2221 and MB2222).

January 4, 2002, letter from C.H. Cruse, The Constellation Energy Group, to NRC, “Calvert Cliffs Nuclear Power Plant Units 1 and 2, Request for Relief from ASME Code Requirements for ECCS and AFW Pump Testing Requirements; PR-12” (TAC Nos. MB3782 and MB3783).

April 17, 2002, letter from P. Salas, Tennessee Valley Authority, to NRC, “Sequoyah Nuclear Power Plant Units 1 and 2, Request for Relief from ASME Section XI Code Requirements, Inservice Testing (IST) Program - Turbine Driven Auxiliary Feedwater Pumps” (TAC Nos. MB4930 and MB4931).

January 22, 2003, letter from M. A. Balduzzi, Entergy to NRC, “Fourth 10-Year Interval Inservice Testing Program and Request for Approval of IST Relief Requests for Pumps and Valves for Vermont Yankee Nuclear Power Station” (TAC No. MB7489).

August 24, 2001, letter from Carolina Power & Light Company to NRC, “Relief Request for Fourth 10-Year Pump and Valve Inservice Testing Program for H. B. Robinson Steam Electric Plant Unit 2” (TAC No. MB2798).

April 15, 2003, letter from Progress Energy to NRC, “Revision to Inservice Testing Program Relief Request IST-RR-3 for Containment Spray Pump Comprehensive Pump Test Requirements for H. B. Robinson Steam Electric Plant Unit 2” (TAC No. MB8447).

United States Code of Federal Regulations, Title 10, Part 50: Domestic Licensing of Production and Utilization Facilities

10 CFR 50.55a, “Codes and standards.”

Federal Register

Regulatory Guide 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code.”

Codes and Standards

ASME/American National Standards Institute (ASME/ANSI), *Operations and Maintenance Standards*, New York, 1987

Part 6 (OM-6), “Inservice Testing of Pumps in Light-Water Reactor Power Plants”

ASME/ANSI, *Code for Operation and Maintenance of Nuclear Power Plants*, 1995 Edition and 1996 Addenda:

Subsection ISTB, “Inservice Testing of Pumps in Light-Water Reactor Power Plants”

Subsection ISTC, “Inservice Testing of Valves in Light-Water Reactor Power Plants”

Code Case OMN-9, “Use of a Pump Curve for Testing”

OVERVIEW OF NRC NUREG-1482, REVISION 1, Guidelines for Inservice Testing at Nuclear Power Plants

Steven M. Unikewicz
Mechanical and Civil Engineering Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Eighth NRC/ASME Symposium on Valve and Pump Testing
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Abstract

The NRC staff is issuing Revision 1 to NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plant," for use by nuclear power plant licensees. Since the initial issuance of NUREG-1482, certain tests and measurements required by earlier editions and addenda of the American Society of Mechanical Engineers (ASME) Code have been clarified, revised or eliminated. The revision to NUREG-1482 incorporates and addresses those changes. The revised guidance incorporates lessons learned and experience gained since the initial issue. This paper provides an overview those changes and discusses how they affect NRC guidance on implementing pump and valve inservice testing (IST) programs. This paper highlights important changes to NUREG-1482, but is not intended to provide a complete record of all changes to the document. Since the issuance of Generic Letter (GL) 89-04, the NRC has improved and clarified its guidance for performing inservice testing of pumps and valves. The NRC intends to continue to develop and improve its guidance on IST methods through active participation in the ASME Code consensus process, interactions with various technical organizations, and through periodic updates of NRC-published guidance and issuance of generic communications as the need arises. Revision 1 to NUREG-1482 incorporates regulatory guidance applicable to the 1998 Edition up to and including the 2000 Addenda to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). It supplements the guidance and positions in GL 89-04. The 1998 Edition up to and including the 2000 Addenda to the ASME OM Code was incorporated by reference into Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(b) and became effective on October 28, 2002 (67 FR 60520). The NUREG document reflects the applicable changes to the paragraph numbering format in the latest OM Code. Revision 0 to NUREG-1482 is still valid and may continue to be used by those licensees who have not been required to update their IST program to the 1995 (or later) Edition of the OM Code. The guidance provided in many sections herein may be used for requesting relief from or alternatives to Code requirements. However, licensees may also request relief or authorization of an alternative that is not in conformance with the guidance. In evaluating such requested relief or alternatives, the NRC uses the recommendations of the NUREG, where applicable. The NRC may reference a recommendation from the NUREG in safety evaluations and grant relief or authorize the alternative if the licensee has addressed all of the aspects included in the applicable section.

Introduction

The U.S. Nuclear Regulatory Commission (NRC) provides licensees guidelines and recommendations for developing and implementing programs for the inservice testing of pumps and valves at commercial nuclear power plants. In NUREG-1482, the staff discusses the regulations; the components to be included in an inservice testing program; and the preparation and content of cold shutdown justifications, refueling outage justifications, and requests for relief from the ASME Code requirements. The staff also gives specific guidance on relief acceptable to the NRC and advises licensees in the use of this information at their facilities. The staff discusses the revised standard technical specifications (TS) for the inservice testing program requirements and gives guidance on the process a licensee may follow upon finding an instance of noncompliance with the Code.

The NRC staff is issuing this NUREG to assist the industry in eliminating unnecessary requests for relief and to provide guidelines and examples acceptable to the staff that might be useful to a licensee considering an alternative IST method to that required in the ASME Code. It is hoped that the guidance in NUREG-1482 will assist the industry in establishing a consistent IST approach. Implementation of the guidance is strictly voluntary and may change depending on advancements in technology or IST techniques. The NUREG also discusses some examples of the use of portions of later OM Code Editions and Addenda that licensees may implement if the related requirements stated in the applicable recommendations are met.

Specifically, the NRC staff is issuing Revision 1 to NUREG-1482 for the following reasons:

(1) To provide guidance on the use of portions of the 1998 OM Code up to and including the 2000 Addenda that the staff has determined are acceptable to implement pursuant to 10 CFR 50.55a(f)(4)(iv). This guidance is generally applicable to the 1995 OM Code including the 1996 Addenda requirements and any differences in guidance are discussed where the Code requirements differ.

(2) To provide guidance on information to be included in relief requests or alternatives in order to ensure a more efficient and effective review and approval by the NRC staff.

(3) To clarify common IST issues that have been identified as a result of NRC inspections, licensees' telephone calls or meetings, public meetings, and NRC staff participation on ASME OM committees.

(4) To indicate the NRC staff's views on the acceptability of or the need for caution in applying certain ASME OM interpretations.

(5) To consolidate references to various documents that apply to IST.

(6) To clarify the information to be included in an IST program, the format for relief requests, alternatives, cold shutdown/refueling outage justifications, and the scope of IST programs.

(7) To clarify the staff's views on certain ASME Code requirements or NRC regulatory positions.

The requirement governing the use of specific ASME OM Code Editions and Addenda is provided in 10 CFR 50.55a. As later Editions and Addenda to the ASME OM Code are incorporated by reference into 10 CFR 50.55a, the NRC staff plans to update NUREG-1482 as needed to reflect the changes in Code requirements or other regulatory positions and criteria.

Background

On April 3, 1989, the NRC issued Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs." It addressed frequently encountered issues such as relief requests, procedural implementation, and technical specification provisions for operability and included 11 technical positions used by the staff in reviewing IST program relief requests and described acceptable alternatives to the Code requirements. The positions in GL 89-04 were not for voluntary implementation in all cases, since the staff requested certain licensees implement the positions of the generic letter.

Since the issuance of GL 89-04, the NRC has recognized the need for more focused regulatory initiatives regarding IST by revising 10 CFR 50.55a and separating the IST and inservice inspection (ISI) programs in paragraphs (f) and (g) of Section 50.55a, issuing specific IST guidance such as NUREG-1482, creating a new regulatory guide for approving OM Code cases, and coordinating with ASME to sponsor periodic symposia on pump and valve issues.

On October 28, 2002, the NRC incorporated by reference into paragraph 50.55a(b)(3), the 1998 Edition up to and including the 2000 Addenda of the ASME OM Code. The OM Code in Subsections ISTB and ISTC specify the IST requirements for pumps and valves, respectively. NUREG-1482, Revision 1 is an update incorporating regulatory changes up to and including the ASME OM Code, 1998 Edition with 2000 Addenda.

When using the ASME OM Code (1995 Edition including the 1996 and 1997 Addenda as well as the 1998 Edition up to and including the 2000 Addenda), the recommendations and guidance in NUREG-1482, Revision 1 essentially replaces the positions in GL 89-04. This document discusses the use of these later Editions and Addenda to the OM Code, which may be implemented by licensees pursuant to 10 CFR 50.55a(f)(4)(iv) and gives guidance for obtaining approval pursuant to 10 CFR 50.55a(f)(4)(iv) when updating an IST program (or portion of the program) to the requirements of a later OM Code.

Discussion

The format of the revised NUREG follows the format of a typical IST program plan (i.e., Development and Implementation, General Guidance, Valves, Pumps, Technical Specifications, Code Non-Compliance, and Risk-Informed Inservice Testing). The Appendices contain a copy of the Nuclear Energy Institute (NEI) White Paper, "Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a," dated September 30, 2002, and a copy of GL 89-04 Supplement 1. The NEI White Paper provides guidance for determining the appropriate regulatory requirement under which a request is submitted to the NRC for approval and sample templates containing the appropriate form and content for preparing a relief request.

Throughout the General Guidance, Valves, and Pumps sections, IST requirements for which licensees have requested relief or proposed alternatives are discussed, and guidance is provided on the type of information that should typically (or in some cases must) be included. They also discuss Code and regulatory issues and provide recommendations and guidance as needed. The discussions of issues and recommendations are not intended to impose additional requirements beyond that required by the Code or the regulations, and, as such, do not represent backfits. Rather, these discussions are intended to clarify existing requirements of the Code or the regulations and may provide recommendations to ensure that Code and other regulatory requirements continue to be met.

Section 2 of NUREG-1482 discusses the development and implementation of an IST program. It describes existing requirements for IST, discusses the scope of an IST program, and provides guidance for presenting information in IST programs, including cold shutdown justifications, refueling outage justifications, and relief requests. The section includes a sample list of plant systems for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) that typically (but not necessarily) contain Code pumps or valves that perform a safety function. The section also includes information needed for licensees to establish the tests and test frequencies proposed for pumps and valves in an IST program.

Two of the more significant changes to this document are the discussion of the use of OM Code cases and the use of the NEI White Paper in the development and submittal of IST programs.

With the incorporation by reference of the OM Code into 10 CFR 50.55a, the NRC staff recognized the need for a new regulatory guide that would approve OM Code cases. This regulatory guide would provide a function similar to that of existing Regulatory Guide 1.147 which approves ASME Code cases applicable to Section XI of the ASME Boiler and Pressure Vessel Code. Accordingly, the NRC staff developed Regulatory Guide (RG) 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code." At the same time, the NRC staff also developed a new Regulatory Guide (RG) 1.193, "ASME Code Cases not Approved for Use." Both of these two new regulatory guides were issued for the first time in June 2003. In Revision 1 to NUREG-1482 the NRC states, "The licensee may implement the Code cases listed in RG 1.192 without obtaining further NRC review, if the Code cases are used in their entirety, with any supplemental conditions specified in the regulatory guide." The following Code cases are listed in RG 1.192 as acceptable to the NRC for application in licensees' OM IST programs:

OMN-2, "Thermal Relief Valve Code Case."

OMN-5, "Testing of Liquid Service Relief Valves Without Insulation."

OMN-6, "Alternate Rules for Digital Instruments."

OMN-7, "Alternative Requirements for Pump Testing."

OMN-8, "Alternative Rules for Preservice and Inservice Testing of Power-Operated Valves That Are Used for System Control and Have a Safety Function per OM-10."

OMN-13, "Requirements for Extending Snubber Inservice Visual Examination Interval at LWR Power Plants."

In addition, the following OM Code cases are listed in RG 1.192 as "conditionally acceptable." These Code cases are acceptable to the NRC for application in licensees' OM IST programs within the limitations described in RG 1.192:

OMN-1, "Alternative Rules for Pre-service and Inservice Testing of Certain Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants."

OMN-3, "Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of LWR Power Plants."

OMN-4, "Requirements for Risk Insights for Inservice Testing of Check Valves at LWR Power Plants."

OMN-9, "Use of a Pump Curve for Testing."

OMN-11, "Motor Operated Valve Risk-Based Inspection Code Case."

OMN-12, "Alternative Requirements for Inservice Testing Using Risk Insights for Pneumatically and Hydraulically Operated Valve Assemblies in Light-Water Reactor Power Plants."

Code Cases OMN-1, OMN-3, OMN-4, OMN-11, and OMN-12 are risk-informed Code cases. Regulatory Guide 1.175, "An Approach For Plant-Specific, Risk-Informed Decision Making: Inservice Testing," describes an acceptable alternate approach for applying risk insights from probabilistic risk assessment (PRA), in conjunction with established traditional engineering information, to make changes to a nuclear power plant's IST program. The approach described in RG 1.175 addresses the high level safety principles specified in RG 1.174 and attempts to strike a balance between defining an acceptable process for developing risk-informed IST programs without being overly prescriptive. Until such time as a risk-informed regulation is promulgated and included in the regulations, the alternative approach described in RG 1.175 must be authorized by the NRC pursuant to 10 CFR 50.55a(a)(3)(i) on a plant-specific basis prior to implementation. Because 10 CFR 50.55a(a)(3)(i) places no restrictions on the scope of alternatives that may be authorized, licensees may propose risk-informed alternatives to their entire IST program or may propose alternatives that are more limited in scope (e.g., for a particular system or group of systems, or for a particular group of components). However, with the issuance of RG 1.192, risk-informed IST methods may be used by licensees without prior NRC staff review and approval. NUREG-1482 further discusses risk-Informed IST in a later section.

NEI issued its white paper entitled, "Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a," dated September 30, 2002. The white paper provides useful guidance in determining the appropriate regulatory requirement under which a "relief request" is submitted to the NRC for approval as well as the appropriate format and content to use in the request. The term "relief request" is used loosely in this instance to denote the various types of submittals to the NRC allowed by 10 CFR 50.55a including alternatives to the regulation [10 CFR 50.55a(a)(3)], impractical relief requests [10 CFR 50.55a(f)(5)(iii)], and requests to use later Code Editions and Addenda [10 CFR 50.55a(f)(4)(iv)]. The NEI white paper has been reviewed by NRC staff, and the staff generally agrees with the format and content in the white paper and encourages its use.

Occasionally, the NRC has receives IST program submittals or partial submittals that lack the start and end dates of the 120-month IST interval or the specific Code Edition and Addenda in use. Some licensees, when developing their IST programs, were not aware that the regulations are issued or updated throughout the year through issuance of Federal Register notices. The Code of Federal Regulations is a codification of the general and permanent rules published in the Federal Register and is kept up to date by the individual issues of the Federal Register. Accordingly, these two publications must be used together to determine the latest version of any given rule. Without this understanding, some licensees mistakenly have used the revision date of the Code of Federal Regulations to determine the appropriate Code Edition and Addenda as required in 10 CFR 50.55a(b) rather than the effective date of the rule as noted in the Federal Register notice. Consequently, a more recent Code Edition and Addenda may have been incorporated by reference in 10 CFR 50.55a(b) as noticed in the Federal Register, which resulted in the program being developed to an incorrect edition of the Code.

NUREG-1482, Section 3 provides guidance and NRC recommendations for several general aspects of IST. The significant changes in clarification and guidance in this section fall into three categories; (1) inservice test intervals/ frequencies, (2) testing at power/on-line testing/entry in limiting conditions for operation (LCOs), and (3) pre-conditioning. With regard to test intervals, the NRC may approve relief for extending a test interval for extenuating circumstances in which (1) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or (2) the system design makes compliance impractical. Impractical conditions justifying test deferrals are those that could result in an unnecessary plant shutdown, cause unnecessary challenges to safety systems, place undue stress on components, cause unnecessary cycling of equipment, or unnecessarily reduce the life expectancy of the plant systems and components. Any requested relief would typically include a technical justification for the deferment. Test interval deferrals and exercise frequencies typically have applied to requests to perform IST cold shutdowns or refueling outages.

Unless accompanied by other acceptable rationale, the necessity to enter into an LCO to perform IST would not be sufficient to justify deferring testing until a cold shutdown or refueling outage. Guidance on issues regarding the applicability of LCO and surveillance requirements has been previously issued by the NRC in GL 87-09. If a licensee chooses to defer testing from quarterly to cold shutdown, or to refueling outages, other justification must be included in addition to entry into an LCO. If the deferral is not justified by additional basis, the licensee must perform tests quarterly, or during cold shutdown (as justified), with entry into the LCO for IST to be completed within the out-of-service time allowed by TS.

Pre-conditioning of structures, systems, and components (SSCs) continues to be an issue of discussion between licensees and NRC staff. In Information Notice (IN) 97-16, "Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing," the NRC staff discussed the longstanding concern regarding unacceptable preconditioning of plant SSCs before testing. The staff noted that experience has demonstrated that some testing cannot be performed without disturbing or altering the equipment. The staff also indicated that any such disturbance or alteration would be expected to be limited to the minimum necessary to perform the test and to prevent damage to the equipment. The staff alerted licensees that, in certain cases, the safety benefit of some preconditioning activities might outweigh the benefits of testing in the as-found condition.

Where the ASME Code does not provide specific provisions related to as-found testing of a pump or valve in the IST program, the staff considers acceptable preconditioning to include such activities as (1) periodic venting of pumps which is not routinely scheduled directly prior to testing but may occasionally be performed before testing; (2) pump venting directly prior to testing provided the venting operation has proper controls with a technical evaluation to establish that the amount of gas vented would not adversely affect pump operation; (3) occasional lubrication of a valve stem prior to testing of the valve where stem lubrication is not typically performed prior to testing; and (4) unavoidable movement due to the set-up and connection of test equipment. In each instance of acceptable preconditioning, the licensee is expected to have a documented evaluation of the preconditioning activity and justification for continued confidence in the IST program to assess the operational readiness of the pump or valve. Unacceptable preconditioning of pumps and valves in the IST program includes such activities as (1) routine lubrication of a valve stem prior to testing the valve; (2) operation of a pump or valve shortly before a test if such operation could be avoided through plant procedures with personnel and plant safety maintained; and (3) venting a pump immediately prior to testing without proper controls and scheduling. Further clarification and guidance is provided in NUREG 1482, Section 3.5.

In an effort to shorten refueling outages, an increasing number of licensees are scheduling maintenance, testing, and surveillance activities while the nuclear power plant is on-line. Several licensees have submitted relief requests to the NRC to conduct inservice testing once per refueling cycle, as opposed to during the refueling outage as required by the Code. The NUREG describes several factors to take into consideration when preparing such requests.

One comment of note is that a risk assessment is often performed to justify taking the SSC out of service. The assessment of risk resulting from performance of maintenance activities as required by 10 CFR 50.65(a)(4) of the Maintenance Rule is not sufficient justification for testing components at power. This assessment is required for maintenance activities performed during power operations or during shutdowns. A risk assessment should address the relative merits of testing at power versus testing during refueling outages.

NUMARC 93-01, Rev. 2, Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, also provides guidance for conducting on-line maintenance and testing that may be useful in planning and conducting on-line activities.

NUREG-1482, Section 4 provides guidance and recommendations on valve issues. Revision 1 addresses check valves, power-operated valves (e.g., motor-, air-, and hydraulically-operated valves), safety and relief valves, and miscellaneous valves such as manual valves and pressure isolation valves. Since the issuance of Revision 0, there have been major changes and developments in the ASME OM Code and IST knowledge, technology, philosophy and methodology. Therefore, the NUREG section on valves was rewritten in its entirety. The complete depth and breadth of the individual changes are too numerous to mention in this paper.

Ongoing issues with regard to check valve categorization, requirements, and test methods are addressed. The current issues and guidance with regard to stroke-time testing of power operated valves are discussed in detail as well as verification of position indication. NUREG-1482, Section 4 also provides guidance on instrumentation and instrument accuracy. The section on relief valves contains only minor changes while guidance with respect to miscellaneous valves such as manual valves and pressure isolation valves should be reviewed for applicability to each plant.

As operating experience with the recent Code changes grows, issues regarding valve IST will continue to emerge and be resolved. The NRC intends to continue to update and improve its IST guidance through participation in standards development organizations and technical groups, issuance of generic communications such as information notices, regulatory issue summaries, and generic letters as well as through regular updates of NRC guidance documents (e.g., NUREG-1482) as the need arises. Revision 1 to NUREG-1482 incorporates generic communications issued up to January 1, 2004. It is recommended that a search of recent communications be performed when evaluating issues regarding valve IST.

NUREG-1482, Section 5 provides guidance and recommendations on pump issues. Revision 1 addresses the use of reference curves, evaluation of pump vibration, the comprehensive pump test (CPT), minimum flow lines, instrument and equipment accuracy, pump drivers as well as other issues of interest in the IST of pumps. Since the issuance of Revision 0, there have been major changes and developments in the ASME OM Code and IST knowledge, technology, philosophy and methodology. Therefore, the NUREG section on pumps was rewritten in its entirety. The complete depth and breadth of the changes are beyond the limits of this paper. However, the CPT and pump drivers will be briefly discussed.

In 1995, OM Code Subsection ISTB introduced a new approach to pump testing wherein pumps were divided into two basic groups, normally or routinely operated pumps (group A) and standby pumps (group B). The Code identifies four type of tests: preservice, Group A, Group B, and Comprehensive tests. Group A and Group B are quarterly tests associated with the pump category (Group A test for Group A pump, etc.). Once every two years, each pump in the program is required to be tested to the more rigorous test requirements of the Comprehensive Pump Test (CPT).

A comprehensive test may be substituted for Group A test or Group B test. Group A test may be substituted for Group B test. A preservice test may be substituted for any inservice test. All pumps would receive a pre-service or baseline test followed by quarterly (periodic) tests. The Code allows the less rigorous pump testing to be performed for certain pumps on a quarterly frequency while requiring a pump test to be performed with more accurate flow instrumentation every 2 years at ±20 percent of pump design flow. The intent is to be able to routinely monitor for degradation using the quarterly test and to verify design capability using the CPT.

The OM Code, ISTB-3300(e)(1) requires that reference values be established within ±20% of the design flow for the CPT. The CPT was developed with the knowledge that there are some pumps, such as containment spray pumps, that cannot be tested at the required high flow rates due to original system design configuration. In these cases, it may be necessary to use the pump's recirculation line for IST. However, recirculation lines are not typically designed ±20% of the design flow.

The NRC may accept the use of a lower flow (reference values less than ±20% of the design flow), as required by Subsection ISTB for the comprehensive test, if the licensee demonstrates to the satisfaction of the NRC in a relief request the impracticality of establishing a reference value within ±20% of the design flow for the CPT. The proposed alternative methods to detect hydraulic degradation and trend degradation must provide reasonable assurance of the pump's operational readiness. The NRC reviews these relief requests on a case-by-case basis.

Pump drivers are outside of the scope of the ASME OM Code with the exception of vibration testing for vertical line shaft pumps where the driver is an integral part of the pump. Most of the pumps are driven by electric motors, which are connected via coupling shafts. Motor vibration due to coupling misalignment may not be realized or measured at the pump. Small changes in vibration of a motor can have significant effects on the pump operation and affect the operational readiness of the pump. While excluded from the ASME Code, the health of pump drivers should be included in a licensee's overall plan for the assessment of its pumping systems.

Issues related to motor drivers of pumps are under consideration by a Working Group Committee (WGC) of the Institute of Electrical and Electronics Engineers (IEEE). IEEE addresses issues related to operations, maintenance, aging and testing of Class 1E equipment in nuclear power plants. The WGC has the task to develop and update the IEEE Standard Criteria for the Testing of Nuclear Power Generating Station Safety Systems.

NUREG-1482, Section 6 discusses revised standard technical specifications. The purpose of a pump or valve inservice test is to assess the operational readiness of the component. Inservice tests are designed to detect component degradation by assessing component performance in relation to operating characteristics when the component was known to be operating acceptably. Thus, the data or information obtained during these tests provide insight into the ability of a component to perform its safety-related function under design-basis conditions until the next test. In contrast, technical specification surveillance requirements typically assess system capability, e.g., the ability of a system or component (e.g., pump) to deliver the flow rate assumed in an accident analysis at the time of the test.

The revised standard Technical Specifications reflect the fact that licensees are required by 10 CFR 50.55a to establish and implement an inservice testing program. Section 6 further discusses this topic and reaffirms previous guidance with respect to Code versus TS test frequencies.

NUREG-1482, Section 7 discusses the process for licensees to follow when a Code nonconformance is found. This section was revised to clarify the relationship between Code and TS noncompliance. The guidance in this section was not significantly changed with the exception of deleting a discussion on Design Bases reviews and including further clarifying guidance on starting points for time periods in TS action statements.

NUREG-1482, Section 8 discusses the development of a risk-informed IST program. This is a new section. In recent years, the potential for a risk-based or risk-informed approach to inservice testing has received much attention and study by both NRC and industry. As of the publication of this paper, only two licensees have risk-informed IST programs, Comanche Peak Steam Electric Station and San Onofre Nuclear Generating Station. The section discusses the regulatory basis for a risk-informed program, the use of risk insights for on-line inservice testing, and the use of ASME OM risk-informed Code cases.

Until such time as a risk-informed alternative to the current Code requirements is incorporated by reference into 10 CFR Part 50, the alternative approach described in Regulatory Guide 1.175 must be authorized by the NRC pursuant to 10 CFR 50.55a(a)(3)(i) on a plant-specific basis prior to implementation. Because 10 CFR 50.55a(a)(3)(i) places no restrictions on the scope of alternatives that may be authorized, licensees may propose risk-informed alternatives to their entire inservice testing program or may propose alternatives that are more limited in scope (e.g., for a particular system or group of systems, for a particular group of components). In either case, the staff expects that the licensee's proposal address the principles described in Regulatory Guide 1.175, including those related to implementation and monitoring.

In an effort to shorten refueling outages, many licensees are trying to do as much maintenance, testing, and surveillance activities as possible with the nuclear power plant on-line. For example, several licensees have submitted relief requests to the NRC to conduct inservice testing once per refueling cycle, as opposed to during the refueling outage as prescribed by the Code. Section 8 discusses several factors to be taken into consideration when preparing (and in evaluating) such relief requests to ensure that the proposed alternative provides an acceptable level of quality and safety. The list is not all inclusive but does provide a useful starting point.

Over the past several years, the ASME has developed a series of risk-informed Code cases related to testing of pumps and valves. When using the ASME's risk-informed Code cases, the testing and performance monitoring of individual components must be performed as specified in the risk-informed component Code cases (e.g., OMN-1, OMN-4, OMN-7, OMN-11, and OMN-12) as modified by any conditions specified in RG 1.192. The use of the Code cases is discussed in both Section 2 and Section 8 of NUREG-1482. The information contained in these sections is not new but, rather, combines information from previously issued sources into one common area.

The ASME Committee on Operation and Maintenance of Nuclear Power Plants (ASME OM Committee) is in the process of developing a new Subsection ISTE of the OM Code that will address risk-informed inservice testing. No guidance with respect to draft ISTE documents are provided in NUREG-1482. Later revisions will address this Code Section once it is approved.

Conclusion

Since the issuance of GL 89-04, the NRC has updated and improved its guidance on performing IST. The NRC intends to continue to revise its guidance as experience is gained and lessons are learned through participation in Code and technical organizations and through regular updates of NRC published guidance as the need arises.

Revision 1 to NUREG-1482 is an update incorporating the most recent regulatory changes including the incorporation by reference of the ASME OM Code, 1998 Edition and the 2000 Addenda. It supplements the guidance and positions in GL 89-04. To the extent practical, it reflects the applicable section, subsection, or paragraph of the appropriate documents (10 CFR Part 50, ASME OM Code, and regulatory guides).

Revision 0 is still valid and may continue to be used by those licensees who have not updated their IST program to the 1995 OM Code (or later).

The requirement for licensees to periodically update their IST programs to later ASME OM Code Editions and Addenda is governed by 10 CFR 50.55a. In the future, NUREG-1482 will be updated on an 'as-needed" basis, as Code requirements evolve or other regulatory changes in direction affect the guidance therein.

References

American Society of Mechanical Engineers/ American National Standards Institute (ASME/ANSI), Code for Operation and Maintenance of Nuclear Power Plants, New York, 1998 Edition up to and including the 2000 Addenda.

U. S. Code of Federal Regulations, Title 10, "Energy," Chapter 1, Part 50, "Domestic Licensing of Production and Utilization Facilities."

Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," April 3, 1989.

NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants, Revision 1," 2004.

NEI White Paper, "Standard Format for Requests from Commercial Reactor Licensees Pursuant to 10 CFR 50.55a," dated September 30, 2002.